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YUCCA MOUNTAIN - SUPPLEMENTAL RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION - SAFETY EVALUATION REPORT, VOLUME 3, CHAPTER 2.2.1.2.1 (SCENARIO ANALYSIS), 4TH SET (U.S. DEPARTMENT OF ENERGY'S SAFETY ANALYSIS REPORT SECTION 2.2.1.2)

- Reference 1: Ltr, Sulima to Williams, dtd 03/03/09, "Yucca Mountain – Request for Additional Information – Safety Evaluation Report, Volume 3, Chapter 2.2.1.2.1 (Scenario Analysis), 4th Set (U.S. Department of Energy's Safety Analysis Report Section 2.2.1.2)"
- Reference 2: Ltr, Williams to Sulima, dtd 02/23/09, "Yucca Mountain – Request for Additional Information Re: License Application (Safety Analysis Report Section 2.2.1.2), Safety Evaluation Report Volume 3 – Postclosure Chapter 2.2.1.2.1 (Scenario Analysis), Fourth Set"
- Reference 3: Ltr, Williams to Sulima, dtd 03/24/09, "Yucca Mountain – Request for Additional Information – Safety Evaluation Report, Volume 3, Chapter 2.2.1.2.1 (Scenario Analysis), 4th Set (U.S. Department of Energy's Safety Analysis Report Section 2.2.1.2)"

The purpose of this letter is to transmit the U.S. Department of Energy's supplemental responses to the Requests for Additional Information (RAIs) identified in reference 1 above. The supplemental responses were developed based on our understanding of the technical areas requiring further clarification, as discussed in a December 1, 2009, public teleconference. During the teleconference, the NRC asked six additional questions pertaining to RAI Numbers 20, 21, 23, 25, 26, 35, 36, and 41. The supplemental responses to two of these questions, one pertaining to RAI Number 23, and the other pertaining to RAI Numbers 35 and 41, are provided as enclosures to this letter. The original response to RAI Number 23 was provided by reference 2 above. The original responses to RAI Numbers 35 and 41 were provided by reference 3 above.

DOE submitted the supplemental responses to the other four questions on December 15, 2009 (two questions pertaining to RAI Numbers 20 and 21) and on January 7, 2010 (two questions, one concerning RAI Numbers 25 and 26, and the other concerning RAI Number 36).



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The DOE references cited in the supplemental RAI responses either have been previously provided with the License Application, as part of previous RAI responses, or are included as enclosures with this submittal. The reference provided with a previous RAI response is identified within the enclosed supplemental response reference section.

There is a commitment in the enclosed supplemental response to RAI Number 23. If you have any questions regarding this letter, please contact me at (202) 586-9620, or by email to jeff.williams@rw.doe.gov.



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OTM:SEG-0238

Enclosures (4):

1. Supplemental Response to RAI Volume 3, Chapter 2.2.1.2, Set 4, Number 23
2. Supplemental Response to RAI Volume 3, Chapter 2.2.1.2, Set 4, Numbers 35 and 41
3. SNL 2008. *Screening Analysis of Criticality Features, Events, and Processes for License Application*. ANL-DS0-NU-000001 REV 00 ERD 04. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20090429.0003.
4. SNL 2008. *Screening Analysis of Criticality Features, Events, and Processes for License Application*. ANL-DS0-NU-000001 REV 00 ERD 05. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20090824.0002.

cc w/encls:

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EIE Document Components:

Enclosure #	File Name	File Size, KB
N/A	001_Trans_Ltr_3.2.2.1.2_Set4_RAI_23_35_41_Sup.pdf	N/A
1 and 2	002_Encl_3.2.2.1.2_Set_4_RAI_23_35_41_Sup.pdf	520
3	003_DOC.20090429.0003_ANL-DS0-NU-000001_R0E4.pdf	1,590
4	004_DOC.20090824.0002_ANL-DS0-NU-000001_R0E5.pdf	1,882

RAI Volume 3, Chapter 2.2.1.2.1, Fourth Set, Number 23, Supplemental Question:

Where in the response to RAI 3.2.2.1.2.1-4-23(b) did DOE provide an alternative validation, or commitment to provide a validation, that demonstrates the benchmark applicability for DOE SNF (except for naval SNF)? The existing analyses have not been validated for the proposed application.

The previously submitted validation was inadequate. The inadequacies concerned the justification for chosen benchmarks, especially for the fresh fuel benchmarks. When the CSNF validation analysis was revised, many of the benchmarks originally chosen were found to be inapplicable using the new methods (e.g., TSUNAMI). Demonstrate that the selected benchmark experiments are similar to the systems to be modeled using a rigorous methodology (such as was used for CSNF validation) for all types of DOE SNF and their degraded configurations.

1. SUPPLEMENTAL RESPONSE

The critical experiments used for benchmarking included, to the extent possible, configurations having neutronic and geometric characteristics comparable to those of the proposed storage package configurations. The fundamental parameters associated with: (1) materials of construction, (2) the geometry of construction, and (3) the inherent neutron energy spectrum affecting the fissionable material were evaluated for similarities and selected accordingly. Fresh fuel benchmarks were used because burnup credit is not being used for DOE spent nuclear fuel (SNF). The approach used for selecting applicable critical benchmarks for calculational methodology validation followed the guidelines listed in NUREG/CR-6361 (Lichtenwalter 1997) and NUREG/CR-6698 (Dean and Tayloe 2001). As discussed in the response to RAI 3.2.2.1.2.1-4-035 (and listed in Table 1 of the response), additional k_{eff} margin was used to account for the potential for an underestimation of the bias due to limitations in the available critical benchmarks, thereby reducing the lower bound tolerance limits used for DOE SNF canisters. Using additional k_{eff} margin to account for uncertainties due to limitations in the geometrical or material representations used in the computational method is consistent with the *Criteria to Establish Subcriticality* presented in ANSI/ANS-8.17.

The criticality feature, event, and process screening analysis is based on the DOE SNF fuel groups being required to remain subcritical (including biases and uncertainties) under fully flooded and degraded conditions. SAR Section 2.2.1.4.1.1.2.2 states: "For DOE SNF, a comprehensive evaluation of various states of degradation from fully intact to fully degraded configurations is performed, and criticality control limits are set based on maintaining subcriticality for the most restrictive degraded scenario, for each criticality DOE SNF fuel group." SAR Section 2.2.1.4.1 presents the methodology and analyses required to confirm that waste forms and canisters are acceptable from a postclosure criticality perspective. Although DOE SNF waste form packaging configurations have not been finalized, the administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application. Criticality analyses are

used to derive loading limits based on a representative fuel for each DOE SNF group. Packaging configurations will be finalized and confirmed to be within the authorized loading limits. If for a particular DOE SNF the derived limits are not satisfied, fuel/configuration-specific criticality analyses and justification of associated benchmarks and margins will be provided before accepting the specific DOE SNF for disposal. Confirmation will be based on demonstrating adherence to fissile loading limits that will be established to ensure subcriticality under the most reactive degraded scenario within the respective DOE SNF fuel group. Benchmarks applicable to these degraded scenarios are limited and may come largely from criticality analyses associated with solution chemistry (originally done to support reprocessing). The applicability of these benchmarks will be evaluated to determine an appropriate bias.

2. COMMITMENTS TO NRC

The DOE commits to update the license application as described in Section 3. The change will be included in a future license application update.

3. DESCRIPTION OF PROPOSED LA CHANGE

The following will be inserted directly after the paragraph in SAR Section 2.2.1.4.1.1.2.4.2 that begins with, “A CL is associated with a specific”

In the case of DOE SNF, the benchmark critical experiments were selected from International Handbook of Evaluated Criticality Safety Benchmark Experiments (NEA 2006b), and included, to the extent possible, configurations having neutronic and geometric characteristics comparable to those of the proposed storage package configurations. Additional k_{eff} margin was used to account for the potential for an underestimation of the bias due to limitations in the available critical benchmarks. Prior to waste receipt, DOE will demonstrate that the bias used in establishing loading limits for DOE SNF canisters conservatively envelopes any uncertainty associated with the limited availability of applicable benchmarks.

In addition, SAR Table 2.2-11 will be updated to reflect the additional k_{eff} margin penalty on the critical limit values as listed in Table 1 of the response to RAI 3.2.2.1.2.1-4-035.

4. REFERENCES

ANSI/ANS-8.17-2004. 2004. *American National Standard, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors*. La Grange Park, Illinois: American Nuclear Society.

Dean, J.C. and Tayloe, R.W., Jr. 2001. *Guide for Validation of Nuclear Criticality Safety Calculational Methodology*. NUREG/CR-6698. Washington, D.C.: U.S. Nuclear Regulatory Commission.

Lichtenwalter, J.J.; Bowman, S.M.; DeHart, M.D.; and Hopper, C.M. 1997. *Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages*. NUREG/CR-6361. Washington, D.C.: U.S. Nuclear Regulatory Commission.

RAI Volume 3, Chapter 2.2.1.2.1, Fourth Set, Numbers 35 and 41, Supplemental Question:

Where in the responses to RAI 3.2.2.1.2.1-4-035 and RAI 3.2.2.1.2.1-4-041 did DOE provide detailed technical justification for the use of zero administrative margin to ensure subcriticality? These RAIs requested one of the following:

- (a) Demonstration that intact and degraded spent nuclear fuel (SNF) configurations will be subcritical in the postclosure period, including the use of a subcritical margin of 0.02.
- (b) Demonstrate that the combined conservatism in modeling assumptions for each configuration exceeds 0.02 Δk_{eff} .

1. SUPPLEMENTAL RESPONSE

The information requested in the RAI was discussed in the response to RAI 3.2.2.1.2.1-4-040 and will be expanded upon here. The analysis of criticality during the postclosure period is probabilistically based on numerous factors such as the probability of waste package misload and probability of waste package breach. As described further in this response, the analysis is not sensitive to administrative reactivity margin in that the other analysis factors are more significant contributors to the probability results. The probability of having one or more criticalities in the repository is based on a sequence of events that must occur to have conditions necessary for the potential of a critical event. If all of these events occur, at least one waste package in the repository system is considered critical (i.e., criticality is assumed without performing a direct k_{eff} calculation).

The only input parameter to the probability of criticality (POC) calculation that has a dependency on the critical limit value is the conditional probability of criticality given a misload. The total integrated probability of one or more criticalities occurring in the repository over 10,000 years is not sensitive to the critical limit (Figure 1) because a misload and the initiating event resulting in a breach of the waste package outer barrier are both independent of exceeding the critical limit. Therefore, analogous to the term “administrative margin” described in FCSS-ISG-10, “The administrative margin is an allowance for any unknown (or difficult to identify or quantify) error or uncertainties in the method of calculating k_{eff} that may exist beyond those which have been accounted for explicitly in calculating the bias and its uncertainty,” excess margin in terms of a probability value has been incorporated into the criticality feature, event, and process (FEP) screening analysis by using conservative and bounding probability values in the probability calculations.

A demonstration of how much excess margin has been incorporated is illustrated in Figure 1 for just one bounding parameter (there are several) in the POC calculation - probability of damage due to seismic vibratory ground motion (P_D). The POC used in the license application for criticality feature, event, and process screening (represented by the red dashed line in Figure 1) conservatively uses the damage frequency to the waste package outer barrier based on a 90% residual stress threshold (RST) to determine the probability of stress corrosion cracking (SCC) breach, within 10,000 years, as a result of seismic vibratory ground motion (P_D). The RST is

expressed as a percentage of the yield strength and represents a threshold value for the stress in the waste package outer barrier when SCC could form. The use of only the 90% RST value in the damage probability estimates is conservative because *Stress Corrosion Cracking of Waste Package Outer Barrier and Drip Shield Materials* (SNL 2007, Table 8-3) indicates that the SCC crack initiation stress threshold criterion for alloy 22 (i.e., the waste package outer barrier material) follows a uniform distribution between 0.9 and 1.05 YS_T where YS_T is the at-temperature yield strength. Therefore, use of the 90% RST maximizes the damage frequency (see Table 1). Using the integrated damage frequency over the 90% to 105% RST distribution to determine the probability of stress corrosion cracking breach (values are listed in Table 1), which is analogous to using the mean of the distribution range as prescribed in NUREG-1804 (NRC 2003, p. 2.2-14), results in a POC illustrated by the green dashed line in Figure 1. Changes to the POC from varying the critical limit (i.e., simulating the use of administrative margin on k_{eff}) over the range from 0.91 to 0.97 are illustrated by the purple line in Figure 1 which has been averaged over RST. As can be seen, the effect of varying the critical limit on the POC is negligible compared to the effect of using a fixed value of 90% for the RST (illustrated with the red dashed line in Figure 1). The use of a damage frequency for 90% RST instead of the average damage frequency over RST results in an increase to the POC of 135% to 170% for critical limits of 0.91 to 0.97, and is bounding over any impact from changing the critical limit (~14% change at a critical limit of 0.91).

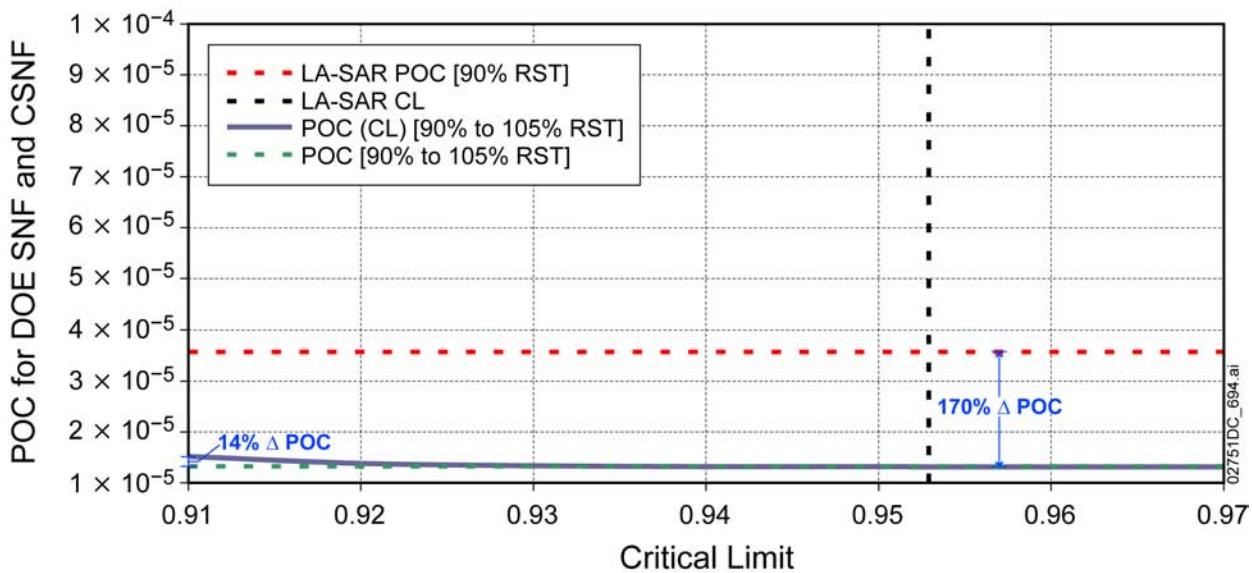
Consequently, because the 10 CFR 63.342(a) requirement for FEP screening is expressed as a probability criterion, a means to provide an allowance for any unknown (or difficult to identify or quantify) error or uncertainties has been incorporated through the use of bounding probability values in the POC calculation in lieu of using an administrative margin on k_{eff} that has a negligible influence on the POC calculation. The effects on the POC are illustrated in Figure 1 by comparing the red dashed line with the purple line and demonstrate that the conservatism used is much larger than the effect of 0.02 Δk_{eff} change on the critical limit.

Table 1. Probabilities of Damage Due to Seismic Vibratory Ground Motion

RST (%)	CDSP Waste Package		CSNF Waste Package	
	Damage Frequency (yr^{-1})	P_D	Damage Frequency (yr^{-1})	P_D
90	2.181×10^{-5}	0.196	1.575×10^{-8}	1.57×10^{-4}
100	4.242×10^{-7}	0.004	0	0
105	0	0	0	0
Expected Value Over Distribution for RST				
90 to 105	7.484×10^{-6}	0.072	5.249×10^{-9}	5.25×10^{-5}

Source: SNL 2008, Table 4.1-6a.

CDSP = codisposal; CSNF = commercial spent nuclear fuel; RST = residual stress threshold; P_D = probability of damage due to seismic vibratory ground motion over 10,000 years.



POC = probability of criticality.

Figure 1. POC for DOE and Commercial SNF as a Function of Critical Limit

Transportation, aging, and disposal (TAD) canister designs and DOE spent nuclear fuel (SNF) packaging strategies have not been finalized. As explained in the response to RAI 3.2.2.1.2.1-4-035 and the supplemental response to RAI 3.2.2.1.2.1-4-023, additional bias on k_{eff} is built in to establishing the DOE SNF canister loading requirements. The amount of excess margin as a Δk_{eff} value was provided in Table 1 of response to RAI 3.2.2.1.2.1-4-035 based on projected loading strategies. The degree of conservatism as a Δk_{eff} value for commercial SNF can be quantified once TAD canister designs are finalized and comparisons against the design basis configurations can be made. However, as explained above, this will have negligible effects on the POC calculation which is the primary metric required by 10 CFR Part 63 for screening FEPs. SAR Section 2.2.1.4.1 presents the methodology and analyses required to confirm that waste forms and canisters are acceptable from a postclosure criticality perspective. The administrative controls described in SAR Section 5.10.2 and SAR Table 5.10-3 require that similar analyses be completed prior to receiving individual waste forms or canisters/waste package design configurations that are not explicitly analyzed in the license application.

During the teleconference on December 1, 2009, the NRC asked for additional clarification on how cross section uncertainty has been addressed. Cross section uncertainty is addressed through the validation process where biases and uncertainties are calculated from critical benchmark experiments. Any effects of cross section error or uncertainties manifest themselves as a bias and are incorporated into the calculation of the critical limit, which is discussed in SAR Section 2.2.1.4.1.1.2.4.2.

2. COMMITMENTS TO NRC

None.

3. DESCRIPTION OF PROPOSED LA CHANGE

None.

4. REFERENCES

NMSS (Office of Nuclear Material Safety and Safeguards) [n.d.]. *Justification for Minimum Margin of Subcriticality for Safety*. FCSS-ISG-10, Rev. 0. Washington, D.C.: Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards. ACC: MOL.20070306.0210.

NRC (U.S. Nuclear Regulatory Commission) 2003. *Yucca Mountain Review Plan, Final Report*. NUREG-1804, Rev. 2. Washington, D.C.: U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards. TIC: 254568.

SNL (Sandia National Laboratories) 2007. *Stress Corrosion Cracking of Waste Package Outer Barrier and Drip Shield Materials*. ANL-EBS-MD-000005 REV 04. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20070913.0001; LLR.20080311.0084; LLR.20080408.0242.

SNL 2008. *Screening Analysis of Criticality Features, Events, and Processes for License Application*. ANL-DS0-NU-000001 REV 00. Las Vegas, Nevada: Sandia National Laboratories. ACC: DOC.20080208.0001; DOC.20080317.0008; LLR.20080401.0255; LLR.20080423.0161; DOC.20090302.0002^a; DOC.20090429.0003; DOC.20090824.0002.

NOTE: ^aProvided as an enclosure to letter from Williams to Sulima dtd 08/11/2009. "Yucca Mountain – Request for Additional Information – Safety Evaluation Report, Volume 3, Chapter 2.2.1.2.1 (Scenario Analysis), 4th Set (U.S. Department of Energy's Safety Analysis Report Section 2.2.1.2) – Resubmittal of Department of Energy Reference Citations."